Fact Sheet
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Next-Generation Reactors

Background

The NRC has long sought standardization of nuclear power plant designs, and the enhanced safety and licensing reform which standardization could make possible. The NRC's regulation (Part 52 to Title 10 of the Code of Federal Regulations) provides a predictable licensing process including certification of next-generation reactor designs. The design certification process provides for early public participation and resolution of safety issues prior to an application to construct a nuclear power plant.

Pre-Application Review Process

The NRC's "Statement of Policy for Regulation of Advanced Nuclear Power Plants," dated July 8, 1986, encourages early discussions, before a license application, between NRC and reactor designers to provide licensing guidance. In June 1988, the NRC issued NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants." This document provides guidance on the implementation of the policy and describes the approach used by NRC in its review of advanced reactor concepts.

The NRC has conducted pre-application reviews of advanced reactor designs to identify:

- major safety issues that could require Commission policy guidance to the staff,
- major technical issues that the staff could resolve under existing regulations or NRC policy, and
- research needed to resolve identified issues.

Design Certification Review Process

The review process for next-generation reactor designs involves the certification of standard reactor designs by a rulemaking process (Subpart B of Part 52). The design certification process requires an applicant to provide the technical information necessary to demonstrate compliance with the safety standards set forth in NRC regulations (10 CFR Parts 20, 50, 73, and 100).

Applicants for design certification must also provide information related to the resolution of unresolved and generic safety issues, issues that arose after the accident at the Three Mile Island plant, a detailed analysis of the design's vulnerability to certain accidents or events, and inspections, tests, analyses, and acceptance criteria.

Currently there are three certified reactor designs that can be referenced in an application for a nuclear power plant application. They are:

1. **Advanced Boiling Water Reactor** (ASWR) design by GE Nuclear Energy (May 1997);
2. **System 80+** design by Westinghouse (formerly ABB-Combustion Engineering) (May 1997); and
3. **AP600** design by Westinghouse (December 1999).

The status of advanced reactor applications for both active and inactive design reviews is provided below in alphabetical order. A description of each design follows.

Reactor Design Review Status

Active Reviews

- **ACR-700** - Atomic Energy of Canada, Limited Technologies, Inc. requested pre-application review of its ACR-700 design in a letter to the NRC dated June 19, 2002. The NRC expects to complete its pre-application review in 2004.

- **AP1000** - Westinghouse requested a design certification review of its AP1000 design, by letter dated March 28, 2002. The NRC expects to complete its design certification review in 2005.

- **ESBWR** - General Electric requested pre-application review of its design in a letter to the NRC dated April 18, 2002. The NRC expects to complete its pre-application review in early 2004 and expects GE to submit an application for design certification in mid-2005.

- **GT-MHR** - On March 22, 2001, General Atomics (GA) requested exploratory discussions with NRC on how to proceed with the licensing of its Gas Turbine-Modular Helium Reactor (GT-MHR) design. NRC will prepare a preliminary technical assessment of the GT-MHR based on the key issues which will need to be resolved as part of an application, GA technical documents, GA white papers, and GA's responses to the NRC's request for additional information and questions. GA has indicated that, at the earliest, a design certification application for the request for additional information GT-MHR would begin by the end of 2005.

• **IRIS** - In an August 12, 2003, letter, Westinghouse outlined its expectations for the near term review of the IRIS design. Westinghouse stated that it is its goal to begin design certification review in 2006, and to deploy the first IRIS module in the 2012-2015 timeframe. Westinghouse requested that the staff conduct a preliminary review of IRIS documents, that a meeting be conducted to discuss the review, and that the staff provide a cost estimate for those near term activities. The NRC staff is presently preparing a response to Westinghouse's August 12, 2003, letter.

• **SWR-1000** - Framatome requested pre-application review of its SWR-1000 design in a letter to the NRC dated May 31, 2002. Framatome intends to begin submitting materials for pre-application review in mid-2004. The NRC expects to complete its pre-application review in 2005.

Inactive Reviews

• **CANDU 3U** - NRC terminated its review at the request of AECL, in March 1995.

• **MHTGR** - NRC discontinued its review in early 1996 at the request of the Department of Energy.

• **PBMR** - Exelon Generation Company, by letter dated December 5, 2000, requested to meet with NRC to discuss issues associated with the potential to license a Pebble Bed Modular Reactor design. An initial public meeting was held January 31, 2001. On April 16, 2002, Exelon announced that it will not be proceeding with the PBMR project beyond the completion of the current feasibility study phase. On May 16, 2002, the staff held a public meeting with Exelon to discuss plans for "wrap-up" of PBMR preapplication review. By letter to Exelon dated September 9, 2002, the PBMR pre-application review was closed.

• **PIUS** - The NRC documented its pre-application review of ABB-CE's Process Inherent Ultimate Safety design in April 1994 and terminated all other activities until an application for design certification is submitted.

• **PRISM** - The Department of Energy submitted the conceptual design for the Power Reactor Innovative Small Module to NRC for pre-application review in November 1986. DOE amended their design document in 1990 and NRC completed its review in February 1994.

• **RESAR SP/90** - The NRC published its final safety evaluation report (NUREG-1413) for Westinghouse's advanced pressurized water reactor design in April 1991 and issued a preliminary design approval. RESAR SP/90 was the first "evolutionary" light-water reactor.

- **SAFR** - The NRC's pre-application safety evaluation report (NUREG-1369) for the Sodium Advanced Fast Reactor design, sponsored by DOE, was published in December 1991.

- **SBWR** - GE Nuclear Energy submitted an application for final design approval and design certification in August 1992. The NRC, in May 1993, determined that it was acceptable for review. In response to some NRC concerns, GE sponsored testing which continued into 1996. However, in March 1996, GE announced the cancellation of the design certification application with an intent to shift the focus of its SBWR programs to plants of 1000 MWe (megawatts electric) or larger. At GE's request, NRC closed out its review activities in early 1997.

Design Descriptions

ABWR: The U.S. Advanced Boiling Water Reactor design uses a single-cycle, forced circulation, boiling water reactor with a rated power of 1300 megawatts electric MWe. The design incorporates features of the BWR designs in Europe, Japan, and the United States, and uses improved electronics, computer, turbine, and fuel technology. The design is expected to show improvement in plant availability, operating capacity, safety, and reliability. Improvements include the use of internal recirculation pumps, control rod drives that can be controlled by a screw mechanism rather than a step process, microprocessor-based digital control and logic systems, and digital safety systems. The design also includes safety enhancements such as containment over pressure protection, passive core debris flooding capability, an independent water makeup system, three emergency diesels, and a combustion turbine as an alternate power source.

ACR-700: The Advanced CANDU Reactor 700 is an evolutionary design adapted from the current CANDU technology. ACR-700 is a 700 MWe class light-water-cooled reactor that incorporates both CANDU and light-water reactor technologies. It uses a conventional CANDU reactor cooling system, with two steam generators and four heat transport pumps. The design uses slightly enriched uranium fuel, separate heavy water moderator and light water coolant, computer-controlled operation, and on-power refueling. The reactor has horizontal pressure tubes supported in a tank filled with the low pressure, low temperature heavy water moderator. The tank also supports the reactivity regulating and safety devices, which are located between the pressure tubes.

AP600: This is a 600 MWe advanced pressurized water reactor that incorporates passive safety systems and simplified system designs. The passive systems use natural driving forces without active pumps, diesels, and other support systems after actuation. Use of redundant, non-safety-related, active equipment and systems minimizes unnecessary use of safety-related systems.

AP1000: This is a larger version of the previously approved AP600 design. It is a 1000 MWe advanced pressurized water reactor that incorporates passive safety systems and simplified system designs. It is similar to the AP600 design but uses a longer reactor vessel to accommodate longer fuel, larger steam generators, and a larger pressurizer.

CANDU 3U: This is a single-loop, pressurized heavy water reactor rated at 450 MWe with two steam generators and four heat transport pumps. The design uses natural uranium fuel, separate heavy water moderator and reactor coolant, computer-controlled operation, and on-line refueling. The reactor has 232 horizontal pressure tubes supported in a tank filled with the heavy water moderator. The tank also supports the reactivity regulating and safety devices which are inserted between and among the pressure tubes. Except for its smaller size and evolutionary design improvements, the CANDU 3U is similar in design to a number of CANDU reactors operating in Canada and other countries.

ESBWR: The Economic and Simplified Boiling Water Reactor (ESBWR) is a 1,390 MWe, natural circulation boiling water reactor that incorporates passive safety features. This design is based on its predecessor, the 670 MWe Simplified BWR (SBWR) and also utilizes features of the certified Advanced Boiling Water Reactor (ABWR). Natural circulation was enhanced in the ESBWR by using a taller vessel, a shorter core, and by reducing the flow restrictions. The ESBWR design utilizes the isolation condenser system for high pressure inventory control and decay heat removal during isolated conditions. After initiation of the automatic depressurization system, low pressure inventory control is provided by the gravity driven cooling system. Containment cooling is provided by the Passive Containment Cooling System.

GT-MHR: The Gas Turbine-Modular Helium Reactor (GT-MHR) design is a 300-MWe helium reactor design based on the high temperature gas-cooled reactor (HTGR) technology. The GT-MHR design uses helium as the coolant and employs refractory fuel. The ceramic-coated particles in the GT-MHR design are contained in fuel compacts that are inserted in graphite fuel elements. The current design allows for up to four 300 MWe modules per common control room. The design is currently being developed jointly by the U.S. and the Russian Federation (under DOE sponsorship) for disposition of weapons-grade plutonium.

IRIS: The International Reactor Innovative and Secure is a pressurized light water cooled, medium power (1000MWt) reactor that has been under development for three years by an international consortium. IRIS is a pressurized water reactor that utilizes an integral reactor coolant system layout. The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant systems components including pumps, steam generators, pressurizer and neutron reflector. The IRIS integral vessel is larger than a traditional PWR pressure vessel, but the size of the IRIS containment is a fraction of the size of corresponding loop reactors.

MHTGR: The Modular High Temperature Gas-Cooled Reactor is a helium-cooled and graphite-moderated thermal power reactor. The fuel is millions of ceramic coated microspheres distributed in cylindrical rods which are inserted in large hexagonal graphite blocks. The blocks are stacked vertically within the reactor vessel through which pressurized helium coolant is circulated. The plant design consists of four identical reactor modules, each with a thermal output of 350 MW, which are coupled with two steam turbine-generator sets to produce a total plant electrical output of 540 MWe. The design includes passive reactor shutdown and decay heat removal features to minimize required reactor operator actions.

PBMR: The Pebble Bed Modular Reactor is a modular HTGR that uses helium as its coolant. PBMR design consists of eight reactor modules, 165 MWe per module, with capacity to store 10 years of spent fuel in the plant (there is additional storage capability in onsite concrete silos). The PBMR core is based on the German high-temperature gas-cooled reactor technology and uses spherical fuel elements.

PIUS: The Process Inherent Ultimate Safe reactor is a 640 MWe advanced pressurized water reactor designed by ABB-Atom of Sweden that utilizes natural physical phenomena to accomplish control and safety functions. The PIUS design consists of a vertical pipe, called a reactor module, which contains the reactor core and is submerged in a large pool of highly borated water. The reactor core is comprised of fuel elements that are similar to current PWR fuel elements. The borated pool water is provided to shut down the reactor and to cool the core by natural circulation. Unlike most reactors, PIUS does not use control rods for controlling the nuclear chain reaction. The reaction is controlled by the boron concentration and temperature of the primary loop reactor water. The steam generating equipment of the PIUS design is similar to that of a typical pressurized light water reactor plant. One important difference in plant design is the very large, by current standards, prestressed concrete reactor vessel. This vessel holds both the reactor module and the borated pool.

PRISM: The Power Reactor Innovative Small Module design uses a modular, pool-type, liquid-sodium cooled reactor. The reactor fuel elements are cylindrical tubes containing pellets of uranium-plutonium-zirconium metal alloy. The reactor uses passive shutdown and decay heat removal features. The standard plant consists of nine reactor modules arranged in power blocks of three reactor modules of 465 MWe. Each module is located in its own below-grade silo and is connected to its own intermediate heat transport system and steam generator system. The steam generator and secondary system hardware are located in a separate building and are connected by a below-grade pipe-way. All the reactors on the site share a common control center, reactor maintenance facility, remove shutdown and radwaste facility, and assembly facility. Each reactor module has its own steam generator which is combined with the two other steam generators in each power block. Total electrical power output would be 1395 MWe.

SBWR: The Simplified Boiling Water Reactor design uses a 600 MWe boiling water reactor with simplified power generation, safety, and heat removal systems to reduce power generation costs, simplify plant safety, and reduce construction times. It uses natural circulation for coolant flow through the reactor. Emergency core cooling is provided by a gravity-driven core cooling system that reduces piping and eliminates pumps and the need for safety-related diesel generators. Because there are no large pipes attached to the vessel near or below the core elevation, the design is intended to ensure full core coverage for all design basis accidents. The isolation condenser is designed as a safety-related system to remove decay heat from the reactor core, by natural circulation and with minimal loss of reactor coolant inventory, following reactor isolation and shutdown.

SWR-1000: The SWR-1000 is a Framatome ANP 1253 MWe boiling water reactor that uses passive safety features. The design is based on a Siemens concept (now Framatome ANP). The SWR-1000 design utilizes emergency condensers to remove heat from the reactor upon a drop in reactor pressure vessel water level. The core flooding system passively floods the reactor core in the event of an accident after the reactor is depressurized. Containment cooling is provided by passive containment cooling condensers. One of the unique design features of the SWR-1000 design is the passive pressure pulse transmitters which provide passive actuation of safety systems.

System 80+: This standard plant design uses a 1300 MWe pressurized water reactor. It is based upon evolutionary improvements to the standard CE System 80 nuclear steam supply system and a balance-of-plant design developed by Duke Power Co. The System 80+ design has safety systems that provide emergency core cooling, feedwater and decay heat removal. The new design also has a safety depressurization system for the reactor, a combustion turbine as an alternate AC power source, and an in-containment refueling water storage tank to enhance the safety and reliability of the reactor system.